3.0 R&D FOR MANAGING PLANT AGING

Degradation of materials in reactor plant structures and components caused by radiation, high temperatures, high pressure cyclic stresses, and a relatively corrosive environment costs utilities that operate light water reactors (LWRs) hundreds of millions of dollars each year. Not only does this degradation increase current operating costs significantly, but, if not managed effectively, material degradation could force the an economic decision to not invest in license renewal for some plants or could prevent cost-effective plant operations to the end of a renewed license term.

For example, one pressurized water reactor (PWR) plant (Trojan) was shut down in part because of the cost of replacing its steam generators (SG), which were damaged by stress corrosion cracking (SCC). Steam Generator replacement cost was a contributing factor to the decision to shut down the Zion plants. SG replacements are expensive, on the order of \$150 million per PWR unit plus the cost of replacement power. As of June 2000, twenty-five PWR plants in the U.S. had replaced their original SGs. For those SGs that do not yet require replacement, repair and inspection as well as operating restrictions in place to lengthen the life of steam generators are significant economic burdens. As another example, extensive cracking has occurred in the core shrouds that surround many of the boiling water reactor (BWR) cores. R&D has already provided many solutions and management strategies for these phenomena. Mechanical clamping systems have been devised to repair core shroud components, but SCC of reactor internals exposed to high radiation (irradiation-assisted SCC – IASCC) continues to be an expensive problem for BWRs. Also, the irradiation of LWR reactor pressure vessels (RPVs) that occurs during reactor operation reduces the ductility of the vessels. Such irradiation embrittlement leads to operational restrictions that can adversely affect the efficiency and life of a plant and its ability to remain economically competitive. In one case, a decision to permanently shut down a reactor (Yankee Rowe) was strongly influenced by questions about RPV embrittlement and its impact on plant economics. The programs that monitor the degree of embrittlement of the RPVs at all nuclear plants in order to ensure that they have adequate ductility are expensive.

Material degradation management for operating reactors requires the ability to predict the rate of degradation as well as the ability to assess the condition of the materials and hence the operability of the affected structures and components. The accuracy and reliability of these assessments control the degree of conservatism that must be assumed in assessing the effects of the degradation and resultant costs. For example, some types of SG tubing defects can be characterized using current nondestructive examination (NDE) techniques and NRC permits continued operation with these safe, characterized defects. When defects cannot be characterized as safe (e.g., by NDE or evaluation limitations), the only two options (both expensive) are to remove affected tubes from service by plugging or repair; for example, by sleeving. Similarly, no NDE technique is accepted generically by the NRC to directly assess the degree of embrittlement of RPVs. Assessments currently rely on empirically developed generic correlations (i.e., based on Charpy V-notch tests) that can be highly conservative when applied to specific plants. New assessment technologies under development such as the Master Curve approach show excellent promise in providing direct measurement of RPV embrittlement, but

more work is needed to reduce uncertainties. Even with the conservative analyses of the old methods, only a few U.S. reactor pressure vessels will face regulatory limitations prior to the end of their current license periods. Master Curve technology promises to resolve RPV issues for all US RPVs through a renewed license term.

Research is needed to understand, characterize, and manage service induced degradation of RPVs, reactor internals, SG tubes, primary system piping, electric cables, and safety-related structures. Technology development needs to be focused on timely detection, mitigation, and prevention of significant long-term effects of aging such as stress corrosion cracking, irradiation assisted stress corrosion cracking, reduction in fracture toughness due to neutron irradiation, thermal embrittlement of cast austenitic stainless steels, piping fatigue, and structure degradation. A research program to address environmental degradation of reactor components, systems, and structures would be a multi-year program involving laboratory tests, component inspections, and technology demonstrations.

This Joint R&D Plan element complements other industry and government R&D activities, and is being conducted in close coordination with other organizations such as NRC. Industrysponsored work is primarily focused on short-term solutions. The ongoing confirmatory research by NRC focuses on issues related to safety. NRC must be assured that assessments of degraded components by the industry are sufficiently conservative to provide adequate safety margins. Said another way, NRC research seeks to characterize the magnitude of the uncertainties involved in the assessment of degraded components, but is not responsible for directly developing technology to reduce these uncertainties. Tight budgets and restricted missions permit relatively little development of base technologies by NRC. For example, the work sponsored by NRC on the NDE of degraded steam generators focuses on the assessment of the capabilities of the current NDE technologies. The development of improved NDE technologies, described in this plan element, is outside the scope of the NRC research responsibilities. Because of its unique facilities and capabilities, DOE is in the best position to address many of the difficult technology issues that involve long-term, high-risk issues. Industry complements DOE capabilities with its knowledge of current plant issues and assists in developing solutions to both short and long-term issues of material degradation.

The research technology areas discussed below are organized by major reactor plant system component. For each major critical component the plan (1) identifies the relevant aging degradation mechanisms and discusses the issues associated with continued safe and economic operation of that component; (2) summarizes the current research and development activities; and, (3) identifies the research needs that remain to be addressed. Volume II provides descriptions of the medium and high priority tasks that have been identified to address the research needs and which require attention in the near-term. Major critical components discussed are reactor pressure vessels; reactor pressure vessel internals; PWR steam generators; reactor coolant system piping, pumps, and valves; electric cables; and structures. Research needs that are not component-specific are also discussed.

3.1 Steam Generators

Steam generators have historically been the single set of components most subject to degradation mechanisms in commercial PWR nuclear plants. As of June 2000, twenty-five plants have already replaced their original steam generators. Such replacements cost on the order of \$150 million per PWR unit. Even for those plants that have not yet had to replace their steam generators, inspection, monitoring, and repair are very expensive, costing nuclear utilities several hundred millions of dollars each year not including replacement energy costs for critical path work activities or due to limitations imposed on the operating plants that result in reduced capacity. Degradation of steam generator tubes has resulted from corrosion and wear, and includes wastage, pitting, denting, SCC, fretting (wear), fatigue, and intergranular attack (IGA). A typical PWR steam generator is shown in Figure 3-1.

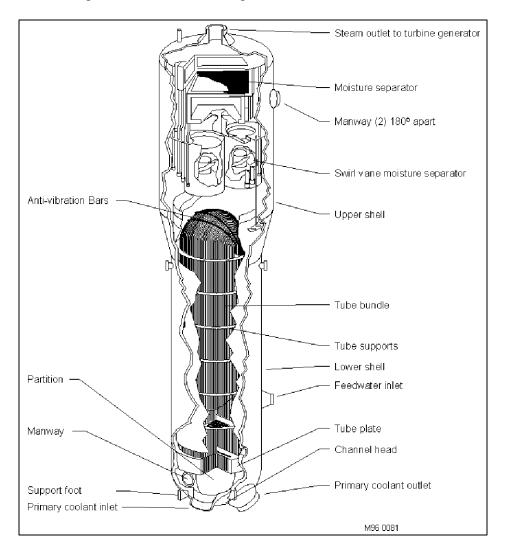


Figure 3-1: Typical PWR Steam Generator

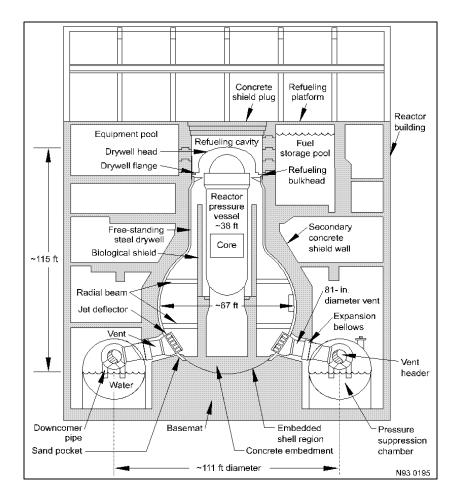


Figure 3-2: BWR Mark I containment (Inverted light bulb and torus)

Locations in steam generator tubes affected by these degradation mechanisms are shown in Figure 3-2. The percent of total number of tubes plugged caused by each of these mechanisms for the years 1973 to 1998 is shown in Figure 3-3. Industry efforts have been largely successful in managing degradation due to wastage, pitting, and denting, but fretting, SCC and IGA have proved more difficult to manage. SCC and IGA are most prevalent in regions where the local chemistries may differ significantly

from those in the bulk such as crevices at the tube sheet and tube support plates, under sludge piles or other deposits, or in regions of high strain and residual stresses such as the roll transition at the tube sheet.

Effective management of the consequences of tube cracking requires (1) nondestructive evaluation techniques to characterize the crack, (2) the ability to characterize the impact of the degradation on the structural integrity of the tubing, and (3) the ability to adequately project the rate of degradation so that the integrity of the tubing can be assured throughout the operating cycle until the next inspection. The accuracy and reliability of the NDE control the degree of conservatism that must be assumed in assessing the effects of the degradation. This strongly affects decisions to repair or replace steam generators. For example, for some types of cracking in steam generator tubing, current NDE techniques provide a sufficiently accurate characterization of the cracking that the NRC permits continued operation of a steam generator in which such cracking has been detected (subject to periodic inspection). For other types of cracking for which no NDE technique has been demonstrated to be adequate to characterize the degradation, the only recourse is to plug or repair the tube upon detection of the crack. An average of about 7,000 tubes per year were plugged or repaired world-wide between 1997 and

1999. Based on the results of tube-pulls and in situ pressure tests, the vast majority of these plugged tubes probably have substantial remaining margin against structural failure.

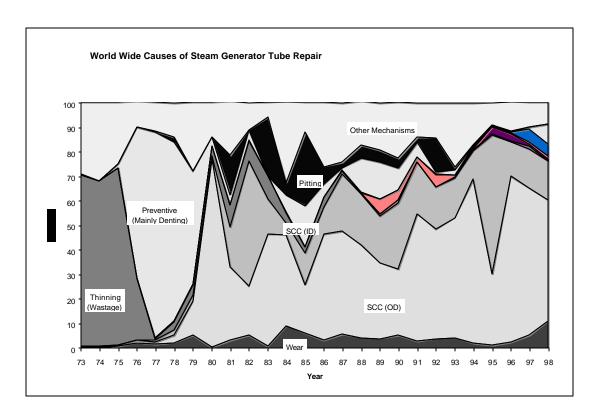


Figure 3-3: Percent of total steam generator tubes plugged due to various degradation mechanisms, 1973 through 1998

If a crack is detected and it cannot be adequately characterized or if the progression of the degradation is too rapid or unknown, the cracked tube must be repaired or plugged. Current repair technologies (mechanical sleeving, and--more recently--electro-sleeving) are very expensive and have not always performed reliably in the past. Plugging is costly and too much plugging degrades the thermal performance of the plant.

The fundamental cause of many of the corrosion problems encountered in steam generators is that the Alloy 600 material used for tubing was not as corrosion resistant as originally thought. The concentration or hideout of bulk-water, low-concentration, low-volatile ionic impurities in local crevice regions, aggravated the problem. Work to mitigate corrosion in current steam generators (i.e., slowing the rate of initiation or progression of the degradation) focuses on understanding and reducing the aggressiveness of the chemistry in crevices. Industry programs to understand the effect of bulk water chemistry on forms of degradation such as wastage and denting have been very successful. However, the relation between bulk water chemistry and the local crevice chemistry is extremely complex (estimates of the concentration factors in crevices

range from 40,000 to 1,000,000), and mitigation of stress corrosion cracking by control of the bulk water chemistry has been less successful to date.

3.1.1 Current Research on Steam Generator Reliability

Both EPRI and NRC are sponsoring research on steam generator degradation. EPRI's work focuses on improved methods of characterizing the current condition of installed steam generators and of predicting and controlling their future rate of degradation to minimize future costs. Results include new and improved technology aimed at residual life management and current steam generator reliability. Results also include collective industry knowledge which can be used in the design, fabrication, installation, and operation of replacement, new, and advanced LWR steam generators. The NRC program emphasizes the assessment of current and emerging NDE technology and the assessment of tube integrity.

EPRI's steam generator R&D is conducted by the Steam Generator Management Program (SGMP), which provides information and technological tools for US nuclear plants to effectively manage steam generator tube degradation, ensure steam generator operability, and reduce maintenance costs associated with steam generators as original equipment or replacement units. SGMP objectives are to further refine and apply technology developed by its ongoing R&D and to refine guideline documents developed in support of the utility industry's Steam Generator Initiative under the Nuclear Energy Institute (NEI; initiative 97-06); to develop QA program plans and procedures for SGMP products; and to manage industry analysis efforts in the resolution of potential severe accident issues as they relate to steam generator operation. EPRI also performs long-term R&D in areas such as water chemistry and thermal/hydraulics that will have significant impact on steam generator operational issues.

NRC's research program focuses on maintaining adequate safety margins for steam generator tube integrity. NRC conducts research to address questions about the nature and effects of steam generator tube degradation mechanisms; effectiveness and reliability of inspection techniques; new repair techniques; and the accuracy of predictive models. Steam generator tubes account for approximately 90 percent of the primary coolant pressure boundary, and failure of these tubes could result in a radiation release directly to the atmosphere. NRC research improves the Commission's and industry's understanding of the nature and effects of normal degradation mechanisms, the effectiveness and reliability of inspection techniques being used today, and about new repair techniques and the accuracy of predictive models. The experimental data provided by NRC's research program provides NRC's licensing office with technical bases for decisions involving steam generator tube cracking behavior, the ability of cracked tubes to withstand normal and accident loads, and the accuracy and reliability of current and advanced inservice inspection methods. This information is used in evaluating licensee programs for managing steam generator tube degradation and in developing generic regulatory positions on acceptable management programs. The outcomes of NRC research relate primarily to the performance goal of maintaining safety by providing the data and analysis tools for the licensing office to independently confirm assessments of steam generator tube integrity.

<u>Improved In-service Inspection and NDE</u>. Current work provides for improved steam generator tubing ISI/NDE performance and the capability to detect, size, and characterize the various

forms of mechanical and corrosion damage to steam generator tubes. This includes evaluating, adapting and integrating various technologies, enhancing existing or current methodologies, and developing new techniques. Also included is the development and maintenance of detailed guidelines on inspection management. Among these are recommendations for inspection sampling and expansion, inspection frequency, requirements for eddy current data analyst and procedure qualification and performance demonstration (PD), and maintenance and updating of the software packages for data analyst training and qualification. One focus will be on assessing the performance of new improved tubing materials used in replacement steam generators.

Improved Steam Generator Materials Performance. Current efforts build on past work funded through the various EPRI-managed steam generator programs since 1977 to develop an understanding of causes and remedies for the materials-stress-environment interactions that have resulted in substantial PWR performance losses due to steam generator corrosion damage. Current efforts also include addressing potential corrosion mechanisms that could affect newer tubing materials (i.e., Alloy 690), and the potential for secondary side material degradation. Deliverables include development of candidate corrosion inhibitors and buffers, development and verification of flaw initiation and growth models, specifications for improved materials and fabrication features, guidance on and specific data relative to tube-pull analyses, documentation on the occurrence and potential causes of specific damage mechanisms, and guidance on repair methods that impact materials performance (e.g., sleeving).

Improved and Advanced Secondary Water Chemistry Guidelines. The goal is a comprehensive set of guidelines, water treatment schemes, software analysis packages, hardware, and processes to control the environmental conditions on the secondary side of PWR steam generators. Deliverables include input to the PWR secondary water chemistry guidelines, input to and development of supporting application guidelines, alternative feedwater pH control additives, secondary cycle chemistry management and analysis software, and sludge removal processes and hardware.

<u>Steam Generator Defect Specific Management (SGDSM)</u>. The objective of the current work is to develop a methodology for and support specific utilities in establishing tube burst and leakage correlations necessary to justify damage-specific tube plugging and repair limits. Deliverables include damage-specific NDE techniques, data analysis and correlation development, and development and maintenance of the SGDSM database.

<u>Steam Generator Technology Support: Status and History Databases</u>. The current effort provides for industry-wide steam generator performance tracking and activities related to information exchange and technology transfer among utilities. Deliverables and milestones include a database and annual progress report on the status of worldwide steam generator performance and degradation progression, revision to the "Steam Generator Reference Book," and topical workshops and information exchange meetings.

<u>SG Tube Thermal Hydraulics</u>, <u>Vibration and Fatigue Workstation</u>. Past work funded through the various EPRI-managed steam generator programs since 1977 has been focused on developing and validating thermal-hydraulic and flow-induced vibration codes. These codes provide for an improved understanding of the operating in-bundle conditions and are used to predict areas of

susceptibility to fouling and vibration damage, to analyze and understand identified T-H, vibration and fatigue problems, and to improve the T-H and vibration design of new and replacement SGs. The results of the current work will include validated codes, improvements to the codes, workstation software and support of utilities in analyzing T-H, fouling, and vibration problems.

The ongoing NEPO projects in this area are:

Project ID: 3-1

Project Title: Steam Generator NDE Test Mockup Facility

Project Description: To provide suitable test specimens and databases of NDE and characterization results that can be used to develop improved NDE techniques and signal analysis methods for the detection and characterization of SCC in steam generator tubing and provide the capability to assess and demonstrate the effectiveness of such methods. A mockup facility will be maintained where defected tubes can be placed in a setting where the artifacts associated with real inspections, such as tube support plates, roll transitions, sludge, magnetite and copper deposits, etc. can be simulated as realistically as possible.

Project ID: 3-2

Project Title: Advanced Eddy-Current Inspection System for Detection and Characterization of Defects in Steam Generator Tubes

Project Description: The nuclear industry has made significant advancements in the development of probes to detect degradation in steam generator tubing. Unfortunately, multiple NDE techniques have been required to achieve optimum degradation detection capabilities. This has lead to increased inspection time. An advanced array probe, consisting of up to 64 individual eddy current coils, will be developed. Such coils can combine rapid inspection speed with high resolution. However, they produce an enormous amount of data and hence, software tools also need to be developed to assist in the analysis of the results.

3.1.2 Additional Steam Generator R&D Needs

Improved NDE Techniques. Improvements in the capability of NDE techniques to characterize defects in terms of their impact on structural integrity are critically needed. Development of advanced NDE systems can draw upon the expertise and experience of the DOE national laboratories, universities, and commercial firms. A variety of approaches to the development of advanced NDE techniques for steam generator tubing can be envisioned. Because there is no clear way *a priori* to identify the best approach to developing improved NDE systems, even though EPRI has ongoing programs, the effort by DOE will be complementary, not duplicative. NRC also sponsors work in this area, but the emphasis of its effort is to assess the capability of the NDE being used by industry to characterize degraded tubing rather than to develop improved NDE technologies.

A major problem in the development of such techniques is the availability of suitable test specimens. Most conventional methods for simulating defects such as electro-discharge machining or even laser cutting techniques produce defects that are poor simulations of the stress corrosion cracks produced in the field. Better, but still not completely representative, cracks can

be produced in the laboratory, but these are still very expensive and can be produced only in a few locations that have the necessary experience and facilities. A significant portion of the effort under NEPO Project 3.27 (described in Section 3.7), *Assessment of Aging Effects on Components and Structures from Nuclear Power Plants*, will be to obtain additional degraded tubing from actual steam generators, similar to the steam generator support plate and tubing material shown in Figure 3-4 from McGuire Unit 1. This tubing, along with laboratory cracked tubing and other simulated degraded tubing, needs to be maintained in a central facility accessible to all interested parties who wish to evaluate techniques for the detection and characterization of degraded tubing. Where feasible the degraded tubing should be placed in a setting where the artifacts associated with real inspections can be simulated as realistically as possible. After the tubing has been characterized by all the NDE techniques of interest, it should be characterized destructively either by metallographic techniques or by leak or burst tests under near prototypical conditions.



Figure 3-4: McGuire Unit 1 steam generator support plate and tubing

The laboratory test facility shown in Figure 3-5 can test degraded tubing under simulated primary and secondary conditions ranging from normal operation to those encountered in a main steam line break. Both axial and circumferential cracks can be tested. The entire progression of the failure from the initiation of a small leak, stable crack growth under increasing pressure, to unstable crack growth, can be simulated in the facility.

<u>Crevice Chemistry and Corrosion</u>. Additional research is needed to better understand the relationship of the bulk water chemistry to the local crevice chemistry. The effect of critical parameters such as bulk water chemistry, degree of crevice superheat, and crevice geometry on crevice chemistry needs to be determined. Instrumentation capable of measuring crevice chemistry conditions at LWR pressures and temperatures needs to be developed. Online

corrosion monitoring systems such as electrochemical noise monitoring that may be suitable for field implementation may provide an alternate approach to assessing crevice chemistry conditions.



Figure 3-5: Argonne National Laboratory test facility for crack initiation and growth testing under simulated reactor operating conditions

<u>Improved Repair Technologies</u>. Current repair technologies are expensive and not always effective. Lower cost, highly reliable repair technologies for steam generator tubes are needed.

Improved Assessment of the Consequences of Tube Leakage or Failure. In assessing the consequences of steam generator tube leaks or failure in terms of the exposure of the public to radiation, the critical radioactive species is iodine. The steady state levels of iodine in reactor coolant systems are very low, but during an accident they can increase significantly. Current regulatory practice requires that the "spiking" factor during an accident be conservatively estimated at 500. More realistic models suggest that the actual spiking factor is less than this, but additional data and investigation are needed to validate these models.

<u>Assessment of Flow-Induced Vibrations</u>. Although corrosion processes are the primary source of tubing degradation, other degradation processes, such as fretting and wear primarily driven by flow-induced vibrations, also occur in steam generators. EPRI has developed flow-induced vibration codes to aid designers in the analysis of these problems. Further work on this subject is needed to make these codes fully effective.

The high priority projects identified for FY 2001 are described below. See Volume II for detailed description of these projects.

Project ID: 3-1 (continuation of current Project 3–1) **Project Title:** Steam Generator NDE Test Mockup Facility

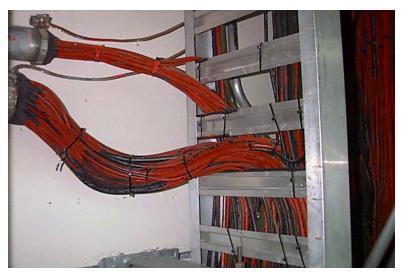
Project Description: This work is a continuation of an ongoing effort which in its first year focused on the development of a library of well characterized, laboratory generated axial, circumferential, ID and OD cracks for use in assessing advanced NDE methods being developed by DOE and EPRI. In the continuation of the project a facility in which tubes from service can be examined will be developed. Artifacts such as tube support plates, tube sheet, magnetite and copper deposits can be simulated in the facility. The primary focus of the work will be on the effectiveness of NDE techniques to characterize cracks.

Project ID: 3-2 (continuation of current Project 3–2)

Project Title: Advanced Eddy-Current Inspection System for Detection and Characterization of Defects in Steam Generator Tubes

Project Description: An advanced array probe is being developed as part of the current Project 3–2. A newly designed transmit-receive array probe has recently been introduced. As in the case of other array probes the quantity of the data acquired can not be efficiently analyzed in a manual mode. Although EPRI is also developing automatic data analysis capabilities for the transmit-receive probe, the industry requirement to use two independent methods for performing data analysis results in the need to develop two different methods for performing automatic data analysis. Alternate approaches may also lead to more effective analysis of the data.

3.2 Cables



Cable systems, consisting of cables, connections and support systems, provide the electrical paths for instrumentation and control devices used for the protection and control of a nuclear plant, and powering of equipment used during normal operation, and in mitigating the effects of accidents. A large quantity of cable is used in a nuclear power plant. A typical new BWR contains ~125 miles of power

Figure 3-6: Vertical Cable Tray

cable, ~600 miles of control cable, and ~10 miles of instrumentation cable. Figure 3-6 shows the transition from a vertical cable tray to a conduit in a typical commercial nuclear power plant, illustrating how large numbers of cables are bundled together.

Cable insulations, fillers, and jackets are organic materials that degrade with time at stress: The higher the stress level, the shorter the time to end of life. In nuclear plant applications, the most common stresses causing degradation are temperature and radiation. In a limited number of applications, these stresses can be amplified by the presence of moisture and chemicals. Cable insulations have been chosen to be able to withstand radiation and thermal stresses for long periods in most applications.

For safety related cables that must operate under accident environments, the insulations must be able to withstand the rigors of normal service while retaining the ability to function through the accident environment. These cables must be replaced prior to the point where ability to function through an accident has been compromised. For cables important to operation, the cables must not fail in normal service. Accordingly, means of evaluating the condition of cable during normal service are necessary for determining the condition of plant cables, especially those in adverse normal environments. Additionally, more accurate models for accelerated aging of cables in environmental qualification programs are needed. With improved accuracy of aging models, prediction of life based on existing qualification test data can be improved.

The environments for most cables are benign and will not cause aging even during a license renewal period. However, most plants have areas with adverse normal environments that can lead to early deterioration. Improving the certainty of aging models used in establishing qualified lives and improving the ability to determine the degree of degradation of cables is critical to assuring that cables needing replacement are replaced and cables with adequate capability are allowed to remain in service.

For low-voltage cables, the most common degradation mode is hardening. This hardening leads to loss of mechanical properties as polymer chains cross-link and simultaneously break into smaller segments. The change in electrical properties during hardening is extremely subtle such that detection by electrical means is difficult. In most cases, degradation of electrical properties is not identifiable until the insulation cracks and moisture enters. Detection of deterioration at this point would be unacceptable because cables would fail in accident environments. Accordingly, most condition monitoring techniques that are available or under development evaluate either a physical or chemical property to determine the degree of degradation and residual life.

If means of distinguishing satisfactory cables from deteriorated cables are not available, plants may have to replace large portions of their cable systems, perform additional qualification testing, and/or develop new programs. An estimate for cable replacement was ~\$80 million per plant. Additional qualification testing for common cable materials, which can only resolve a subset of the aging issues, is estimated to cost ~\$2-5 million. Recent US Nuclear Regulatory Commission research (i.e., program for resolution of GSI-168) indicates that re-qualification will not result in longer qualified lives at elevated temperatures. The cost of qualification testing for plant unique cable materials would be several times the cost of qualifying the common materials

because many different materials were used in early U.S. nuclear plants. Because requalification is expected to provide little benefit and will be expensive, cable life must be evaluated with respect to actual service conditions. Normal environmental stresses must either be shown to be benign, or replacement of cables must be implemented. The replacements may be based on conservative analysis or actual physical condition. Given these two options, two basic areas of research are necessary: development and confirmation of aging models, and development of condition monitoring techniques. For success of this program, a cable condition monitoring database and cable system assessment training must also be developed.

3.2.1 Current Research on Cables

Aging Assessment and Prediction:

Current EPRI research related to aging of nuclear power plant cables and assessment of their condition includes a natural vs. artificial aging project and ionizable gas testing research. The natural vs. artificial aging project is determining material-specific correlations between in-plant natural aging and accelerated aging practices by evaluating cable samples that have been placed in nuclear power plants for long durations. Bundles containing cable specimens were placed in power plants in 1985 and are periodically removed for evaluation. Valuable data on the condition of common cable types is being developed, along with a basis for validation of aging models.

With the development and verification of aging models and condition evaluation techniques, a training program is necessary to provide the results in a form that is useful to plant personnel. The training program will inform plant personnel of means of analyzing overall cable system condition and evaluating the physical condition of cables systems by means of simple manipulative techniques through sophisticated condition monitoring techniques capable of identifying residual service life.

Development of Empirical Data to Characterize Aging Degradation of Polymers Used in Electrical Cable:

As plant-operating experience accumulates and additional research is performed, aging behaviors that could not be predicted from early experiments and environmental quality (EQ) testing are being identified. Figure 3-7 shows highly irradiated cables (lower right of photo) and new cables (upper left of photo) from one of the Department of Energy Savannah River plants. Although the insulation damage at Savannah River was more severe than expected at a commercial plant, and most currently installed cable materials will perform their safety function for many years, uncertainties related to long-term cable system operability and Arrhenius methodology calculations of qualified life values are a significant engineering and regulatory concern. A science-based understanding of polymer degradation is needed to address safety concerns, and identify and resolve long-term operational issues.

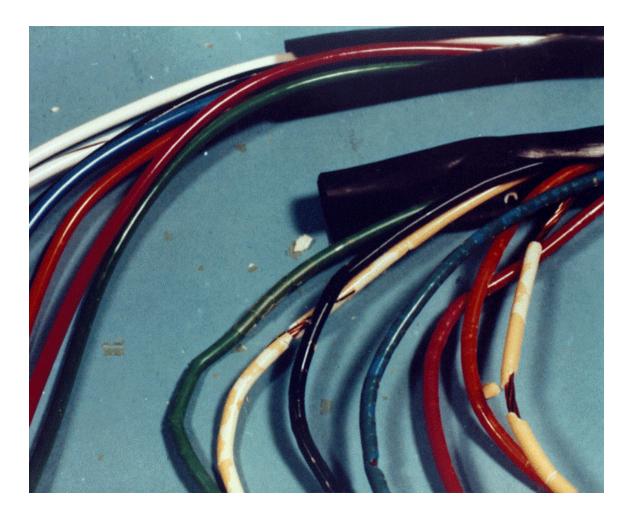


Figure 3-7: Savannah River low density polyethylene cables

Natural Aging Compared to Accelerated Aging Predictions. One funded task attempts to compare natural aging to model predictions based on accelerated aging. This involves an EPRI/University of Connecticut (UConn) project that was initiated in 1985 to establish a material-specific correlation between natural (*in situ*) aging and accelerated aging for four commonly used cable types. An additional program was initiated in 1999 in which cable specimens are being subjected to long-term, low level thermal aging. The acceleration factor for this program is 10 to 15 times normal rates, which is exceedingly low by comparison to the 700 to 2000 factors used in environmental qualification. This low-rate aging program will provide long-term (e.g., end of license renewal period) aging information that will be useful in understanding expected cable system degradation.

<u>Plant Operational Cables</u> - Cable samples from decommissioned plants (e.g., Yankee Rowe), cable replacement activities (e.g., Big Rock Point, before the August 1997 shutdown), and cable monitoring programs (e.g., Oconee) have been and will continue to be gathered. EPRI/UConn

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Three of the types manufactured by Okonite, BIW, and Kerite have ethylene-propylene copolymer (EPR) insulation and chlorosulfonated polyethylene (CSPE)/Hypalon jacket, and the fourth type manufactured by Rockbestos has cross-linked polyethylene (XLPE) insulation and Neoprene jacket.

cables are not electrically connected; however, they are in the same external environment as plant cables.

Obtain Naturally Aged Samples. Materials from nuclear power plants and DOE facilities are needed to cover commonly used cable materials. The Low Intensity Cobalt Array (LICA) facility at Sandia (shown in Figure 3-8) can subject cable samples to long term radiation exposure with electrical excitation and appropriate thermal and gas environments. As additional materials and the corresponding environmental information are gathered, the samples will be evaluated and data placed into the condition monitoring database.



Figure 3-8: Low Intensity Cobalt Array (LICA) for radiation and thermal aging

Confirmation and Development of Aging Models. Uncertainties related to long-term cable system operability and Arrhenius methodology calculations of qualified life are a significant regulatory concern. Older approaches use short-term, high temperature elongation at break data to derive Arrhenius activation energy (E_a) and then extrapolate to service conditions assuming E_a is unchanged. This task will use ultra-sensitive O_2 consumption (UOC) measurements at high temperatures to show the correlation with elongation and continue the UOC measurements down to service temperatures to quantitatively test the Arrhenius extrapolation assumption. Material/environment-specific experiments will identify when the Arrhenius equation works and when an alternative model is needed. Experiments will be conducted on cable materials for which the time- and temperature- dependent elongation results already exist as well as on other newly procured materials where elongation measurements will follow oven aging. Conclusions about generic behaviors for important cable materials (e.g., Hypalon cable jackets) should be available, which would significantly reduce cable issues.

Investigate Bonded Jacket Cable Failure Mechanisms. A US NRC qualification research program for resolution of GSI-168 confirmed a failure mechanism associated with bonded jacket cables where cracks in the jacket propagate through the insulation. The failure mechanism relates to over aging of the bonded Hypalon™ jacket. When the jacket is exposed to loss-of-coolant accident environments (LOCAs), splitting of the jacket can occur. The split propagates through the jacket and insulation to the conductor. Research is necessary to identify the condition to which the jacket may be aged without causing LOCA environment induced failure. The resulting information will allow use of the cables within existing environmental qualifications.

Evaluation of the "Wear-out" Methodology. In this method, pieces of material that have aged for long periods of time under ambient field conditions are subjected to an accelerated "Wear-out" temperature to drive the material to its "failure" condition. For constant acceleration conditions (the Arrhenius assumption), a linear relationship is predicted to occur between real time aging exposure and the time to failure at the Wear-out temperature. A plot of this relationship allows an estimate of material lifetime under ambient conditions, offering an alternative means of periodically checking the predictions available from the accelerated extrapolations. The method should be particularly useful for the many cable materials that show little evidence of damage with aging time until they catastrophically fail (referred to as "induction-time" behavior), since the Wear-out approach should transform such non-predictive behavior into linear predictive behavior. predict remaining lifetime.

Development of Condition Monitoring (CM) Techniques and Data for Electrical Cable:

• Condition monitoring can be used to assess the current condition of a cable and, with evaluations based on empirical data, provide assurance that a cable has sufficient remaining life to perform its safety function(s). Condition monitoring (CM) methods that measure a mechanical, physical, or chemical property at one or more specific locations in a cable run (i.e., sampling techniques) are needed to provide data that quantify the local mechanical condition of insulation and jacket materials. From experience and numerous tests, a threshold mechanical condition has been associated with electrical functional capability. While an electrical test is desirable to assess cable insulation degradation, to date, mechanical property testing through in-situ or sampling tests have been developed to identify precursor mechanical conditions (e.g., insulation hardening prior to cracking) that will ultimately detract from electrical functionality. Significant research is still needed to allow detection of insulation through evaluation of the subtle changes to electrical properties.

<u>Development of Cable System Assessment Techniques.</u> Sensory (i.e., visual and tactile) inspection can identify anomalous cable and termination conditions. A comprehensive cable system assessment-training program needs to be developed for use by plant personnel. The program should include:

- Means of identifying cables at risk of degrading prior to end of plant life
- Use of aging models for cable system assessment

- Use of existing cable CM data
- Use of visual/tactile assessment of in-service cables
- Integration of cable assessment techniques into on-going maintenance and surveillance activities
- Determination of frequency and level of assessment
- Determination of need for replacement or implementation of more sophisticated CM assessment techniques

The training program must be assessed and refined by pilot use in an operating plant.

<u>Develop a Comprehensive CM Data Set</u>. Data collected from other tasks needs to be reviewed to identify any gaps in baseline data for commonly used cable materials. Field use of NDE techniques based on oxidation induction time (OIT), oxidation induction temperature (OITemp), density, indenter, and additional CM techniques developed under the Nuclear Energy Plant Optimization (NEPO) program needs to be based on a comprehensive collection of test data. As required, material-specific samples need to be artificially aged and tested to cover the required range of environmental conditions. After analyzing the test data, additional inputs need to be incorporated in other cable tasks.

<u>Develop Sample Removal Techniques</u>. Nuclear power plants need micro-sample removal techniques that do not affect cable qualification while ensuring that representative samples are collected and cataloged. Sampling techniques and tooling need to be developed and qualified. TR-112233, *Polymer Specimen Removal Techniques for In-Service Cables*, February 2000, provides basic methods for specimen removal and insights into the degree of difficulty of specimen removal. Without further research, specimen removal from active cables will require repairs or retermination of conductors.

Investigate Additional Advanced CM Techniques (Modulus Profiling, Density, and NMR). The goals of this task include a careful and systematic investigation of the correlation of elongation results with data taken from several promising CM techniques including modulus profiling, density and nuclear magnetic resonance (NMR). Each of these CM approaches offers the potential for making measurements on very small samples (< 1 mg), much smaller than the size required for other candidate CM techniques (OIT, for instance, requires ~8 mg). In addition, the small sample requirements means that the CM samples could also be used as a way of estimating residual cable lifetimes using the Wear-out method described above. For materials and environments where a particular CM approach appears to be promising, further studies will be conducted to determine how the correlation depends on environmental stress level (e.g., on the temperature or radiation dose rate). This is critically important since the accelerated stress levels where the correlation is derived will be much higher than the stress levels appropriate under the ambient aging conditions of interest to the eventual application of CM techniques.

<u>CM Database</u>. Cable CM data (e.g., indenter modulus, density, and oxidation induction time) from research organizations, universities, and utilities need to be compiled for use in comparisons to cables in the field or removed from service. CM data needs to be stored in an electronic format designed to facilitate use by plant engineers and researchers. The database will

promote consistency in measurement techniques, data collection and interpretation of laboratory and field results.

The ongoing NEPO projects in this area are:

Project ID: 3-7

Project Title: Develop Empirical Data to Characterize Aging Degradation of Polymers Used in Electrical Cable

Project Description: Develop empirical data to characterize the aging behavior of polymer materials in electrical cable insulation and jackets for the following environments; typical power plant conditions, R&D laboratory experimental conditions, and environmental qualification tests. Studies testing the Arrhenius model (using oxygen consumption) will initially concentrate on materials from SCRAPS (Sandia's Cable Repository of Aged Polymer Samples) for which long-term aging results (tensile properties) are already available. For these materials, much of the necessary consumption data to test the extrapolation assumption will be obtained under this task.

Project ID: 3-8

Project Title: Develop Condition Monitoring (CM) Techniques for Electrical Cable

Project Description: Develop nondestructive or essentially-nondestructive, science-based, CM techniques for electrical cable insulation and jacket materials that are capable of characterizing the current condition of either a local section or an entire cable run using parameters (e.g., density) correlated to aging models or other well-defined criteria. Confirmation of ability of identification of damaged insulation via electrical testing in an ionizable gas environment and extension ionized gas testing to locate and characterize damage sites will also be included. Development of a distributed temperature and radiation monitoring system based on fiber optic transmission characteristics will also be evaluated. Extensive screening of modulus profiling and density measurements and initial screening of nuclear magnetic resonance (NMR) and other promising CM techniques using SCRAPS (Sandia's Cable Repository of Aged Polymer samples) samples will be performed under this task.

3.2.2 R&D Needs on Cables

Most of the available condition monitoring techniques are based at point evaluations of insulation and jacket materials. They must be applied with a knowledge of the location of the most severe environment. In some cases, the location of the severe environment is not readily identifiable. Therefore, techniques that evaluate the entire length of the cable are desirable. The one technique that could allow cable evaluation in this manner is electrical testing. While known to be difficult, advancements in detection systems and computer assessment of data may allow ultimate use electrical testing of unshielded, low-voltage cable. In addition, sample removal technique improvement is necessary for all of the available laboratory tests relying on evaluation of 1 to 8 mg of insulation or jacket material. The following list describes some of the programs that may be implemented at a future date.

Medium Voltage Cables Medium-voltage (4 to 15kV) cables provide power to non-safety and safety-buses, large motors for emergency and operations pumps necessary for mitigation of accidents, and protection and control of a nuclear reactor. Management of aging of cables is necessary to assure satisfactory function of cables under normal and accident conditions. Unlike low-voltage cable that deteriorate primarily due to thermal and radiation stresses, medium voltage cables also deteriorate from electrical stress on the insulation. Research is needed to understand aging rates, and how to detect and monitor degradation. Use of appropriate monitoring technology will reduce the need for early replacement of cables and assure that aged cable is replaced before it becomes a problem.

<u>Develop Electrical NDE Techniques</u>. An electrical NDE technique capable of detecting and locating incipient defects in an entire cable run is desirable. Current mechanical tests may only be implemented at points along a cable run where the surface of the insulation or jacket are exposed to view (e.g., not contained in a conduit). To date, changes in low-frequency tangent delta have been shown to correlate with aging. However, successful measurements have only been made on shielded cables under laboratory conditions. Research is needed to develop a robust electrical test technique for use on non-shielded cables in actual plant conditions.

Determine Whether Electrical NDE Techniques Are Nondestructive. Cables used to perform CM tasks need to be destructively examined to ensure that they are not damaged by electrical NDE techniques. High potential ("continuous" and pulse), partial discharge, and insulation resistance electrical inspection techniques need to be investigated to determine whether conditions just below the fault threshold may damage the polymer. Research is needed to determine whether electrical NDE techniques degrade the insulation and/or jacket of a cable system.

<u>Develop Models for Electrical Breakdown Phenomenology</u>. At present, there are no theoretical models for predicting electrical breakdown due to defects in polymers. Experts in the field believe it may be possible to develop a combination of analytical tools and test data to characterize the electrical behavior of polymer defects. Research is needed to provide a basis for acceptance criteria used with electrical NDE techniques. This activity is applicable to medium voltage cable used in nuclear plants

<u>Development of a Systems Based Approach for Condition Monitoring of Inaccessible</u> Instrumentation and Control Cables

This project will develop a non-destructive, systems level condition monitoring technique for instrumentation and control cables that is applicable to accessible, as well as inaccessible cables

The high priority projects identified for FY 2001 are listed below, along with a brief project description. See Volume II for detailed descriptions of these projects.

Project ID: 3-7

Project Title: Develop Empirical Data to Characterize Aging Degradation of PolymersUsed in Electrical Cable (continuation)

Project Description: This work is a continuation of an on going effort which in its first year focused on studies testing the Arrhenius model using oxygen consumption on materials from laboratory aged samples. Under this task, other important cable materials (long-term-ambient aged cable samples from nuclear power plants) will be acquired to develop empirical data to characterize with higher certainty the aging behaviors and remaining lifetimes of polymer materials in electrical cables used in nuclear power plants.

Project ID: 3-8

Title: Develop Condition Monitoring (CM) Techniques for Low-Voltage Electrical Cable (continuation)

Project Description: This work is a continuation of an on going effort which in its first year focused on screening of CM techniques using Sandia's Cable Repository of Aged Polymer samples (SCRAPS). Under this task, Concentration will be on developing and refining the CM techniques found to be most useful for each generic cable material with special emphasis on optimizing each method for minimum sample size. Also the CM data will be expanded on important newly acquired cable samples that first require aging before CM evaluation.

Project ID: 3-8A

Title: Develop Condition Monitoring (CM) Techniques for Medium-Voltage Electrical Cable

Principle Objective: To identify the configurations of cables in-service and to link them to existing electrical CM technologies

3.3 Reactor Internals

Intergranular stress corrosion cracking (IGSCC) in BWR piping systems, which was a significant problem in the late 1970's, involved "sensitization" of the stainless steel, i.e., depletion of the chromium at the grain boundaries. Neutron irradiation over time can result in susceptibility to IGSCC even in nonsensitized stainless steels because of the redistribution of the impurity and alloying elements at the grain boundaries, and embrittlement of the material. This irradiation induced IGSCC is referred to as irradiation assisted stress corrosion cracking (IASCC). Figure 3-9 shows the increasing predominance of IGSCC on BWR neutron absorber tubes from an operating reactor with increasing neutron fluence.

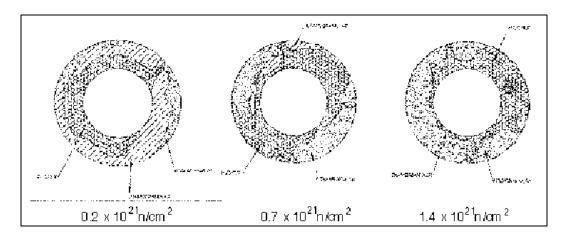


Figure 3-9: Intergranular stress corrosion cracking of BWR neutron absorber tubes as a function of neutron fluence

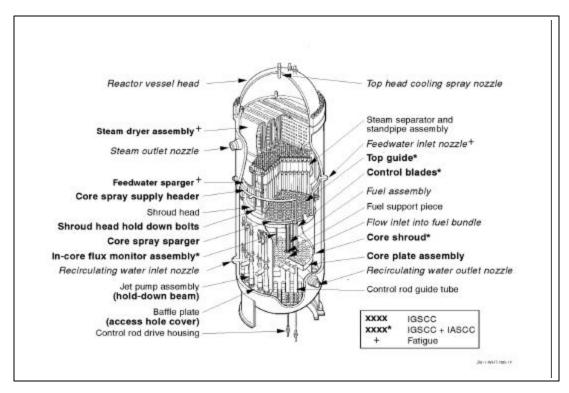


Figure 3-10: Cutaway view of BWR reactor pressure vessel and internals, showing location of areas susceptible to stress corrosion cracking and fatigue damage

High radiation levels in a reactor core can also increase the susceptibility of the core structural materials to stress corrosion cracking because of changes in the water chemistry due to the radiolytic decomposition of the water and degradation of the materials themselves. The higher levels of aggressive radiolysis products such as hydrogen peroxide (H₂O₂) increase the electrochemical potential of the coolant. The arrangement of the control rod drive and reactor internals used in the BWR-3 and BWR-4 designs is shown in Figure 3-10. IGSCC, IASCC, and fatigue damage locations are identified in the Figure. Although the PWR water environment is

less aggressive than that in BWRs, some PWR core components are exposed to even higher fluences and may also become susceptible to failure, although the actual microstructural mechanisms may be somewhat different than in the case of the BWR. Some internals in PWRs are subject to fluences much higher than those in BWRs. Such components could be subject to other mechanisms of degradation, such as swelling, in which an unconstrained component could increase substantially in size. Such behavior has been widely observed and studied in fast reactors, which typically operate at higher fluences and higher temperatures than LWRs. The potential for such behavior in PWRs is unclear.

The major issues related to managing plant aging for the reactor internals are: (1) managing the effects of IGSCC for susceptible internals components (e.g., austenitic stainless steel material, a crevice- containing geometry, and sensitized material microstructure) that are exposed to reactor water with sufficiently high electrochemical potential; (2) managing the effects of IASCC for internals components exposed to sufficiently high neutron flux over a sufficiently long time to reduce the material ductility and fracture toughness to levels of concern; (3) managing the effects of stress relaxation for those internals components (e.g., bolts, pins, and fasteners) that depend upon preload to maintain their function, and (4) addressing potential issues such as swelling. Other plant aging issues, such as cracking caused by high-cycle fatigue, wear caused by relative motion between parts, and dimensional changes caused by exposure to radiation, are well understood and are of less significance.

The current approach for managing the potential effects of IGSCC, IASCC, and stress relaxation is based on periodic in-service inspection (e.g., visual examination) of internals components that can be removed from the reactor vessel, or internals components that are accessible or can be made accessible for examination, in order to detect any relevant conditions. This approach also includes supplementary examinations to further characterize relevant conditions, plus any corrective actions or engineering evaluations intended to demonstrate fitness for continued service.

3.3.1 Current Research on Reactor Internals

Current R&D on reactor internals is directed at condition assessment (e.g., in-service inspection and engineering evaluation of relevant conditions) of components in operating plants, prevention or mitigation of the service conditions that might lead to degradation, and development of replacement materials that are less susceptible to degradation.

Condition Assessment Guides for BWR Internals. BWR owners have organized a collaborative effort, the BWR Vessel and Internals Project, (BWRVIP) to develop methodologies to manage the degradation of reactor internals. The BWRVIP is developing a comprehensive set of guides that will provide utilities with the information needed to make cost-effective decisions to manage degradation of the BWR reactor vessel and internal components. These condition assessments will cover in-service inspection procedures, repair or /replacement techniques, potential mitigation methods (e.g., water chemistry guidelines), and overall assessment of fitness for continued service.

Engineering Evaluation of IGSCC. R&D on crack growth rates is being performed to support assessments of serviceability and service life when SCC is indicated by inspection, to avoid unnecessary or premature repair/replace decisions and to quantify the benefits of countermeasures such as hydrogen water chemistry (HWC) or noble metal chemical addition (NMCA). Three separate initiatives have been taken to collect and correlate data on stainless steels, on nickel-base alloys, and on low alloy steels, respectively. The NRC is also supporting work to determine IASCC crack growth rates.

Stress corrosion is influenced by residual stresses, which are typically high at weldments. BWRVIP has supported residual stress measurements and analyses on typical welded construction of key BWR internals. BWRVIP also supports development of analytical models predicting the effects of gamma and neutron fields on chemical dissociation and recombination in BWR coolant circuits. And, the BWRVIP has continued prior EPRI developments on processes and consumables for repair welding, including underwater welding.

Mitigation Methodologies. Since the early 1980s, BWR plants have had the option of adding hydrogen to the reactor coolant to suppress stress corrosion cracking. The noble metal chemical addition (NMCA) process developed by General Electric produces a very thin durable deposit of platinum and rhodium on reactor internal components that improves the effectiveness of hydrogen water chemistry by enhancing the recombination of radiolytic oxygen with injected hydrogen. The application of NMCA was demonstrated by the BWR VIP in the Duane Arnold BWR in October 1996. Ongoing work includes monitoring of electrochemical potential and crack growth over two cycles and fuel surveillance over three cycles. The NMCA fuel surveillance plan also included post-irradiation examination of discharged fuel in a hot cell to look for evidence of increased hydrogen uptake in the fuel cladding which could be a precursor to hydriding or hydrogen embrittlement. No fuel degradation has been observed at Duane Arnold. NMCA has been applied at fourteen other US BWRs, and will be applied to twelve more BWRs in the US and abroad in the near future. Water-jet peening to mitigate IGSCC in BWR and PWSCC in PWR plants has been investigated under a DOE small business innovative research contract.

<u>IASCC</u> Research. The Cooperative IASCC Research program (CIR) is a five-year program managed by EPRI to develop a mechanistic understanding of IASCC, to develop a predictive methodology based on this understanding, and to identify possible countermeasures to IASCC. Collaboration is motivated by the high cost of research on irradiated materials. International sponsors of the CIR program include the NRC, other foreign regulatory authorities, NSSS vendors, nuclear utilities and utility organizations. A current focus is collection and analysis of existing laboratory and field data related to IASCC. A database has been developed and is periodically updated. An in-depth literature study has been published. Key technical issues have been identified to guide the research. Work is in progress on a systematic in-depth study of irradiated materials known to be either susceptible or resistant to IASCC. Materials are being tested in both aggressive and benign environments found in both BWRs and PWRs. Critical testing parameters and acceptance criteria will be established for qualification of replacement materials.

Stress Corrosion Cracking in PWR Alloy 600 Penetrations. The Nuclear Energy Institute (NEI) has organized an ad hoc task group to resolve issues associated with stress corrosion cracking in PWR Alloy 600 vessel head penetrations. EPRI has a multi year research project that began in 1993 to provide data and assessment methods to predict the residual life of Alloy 600 penetrations in PWRs, including crack growth rates as a function of microstructure, stress and water chemistry, and the effect of microstructure on crack initiation in penetration materials and evaluate remedies such as shot peening and zinc additions to the PWR primary water.

Procedures and hardware for laser weld repair of PWR vessel head penetrations damaged by stress corrosion cracking are also being developed.

Replacement Material for Baffle/Former Bolts. PWR core internals include structures called baffles and formers that surround the reactor core (shown in Figure 3-11). The baffle/former bolts that hold these structures together are exposed to high neutron radiation levels, and have failed in a number of plants in the U.S. and overseas. In conjunction with the three owners groups for PWR operating reactors in the US (WOG, B&WOG and CEOG) EPRI is supporting programs relevant to operating PWRs through the PWR Material Reliability Project (MRP). EPRI is also supporting research and development programs in progress at Electricite de France (EDF) that are directed at determining the effects of irradiation on the mechanical and stress corrosion behavior of the materials used in baffle/former bolts.

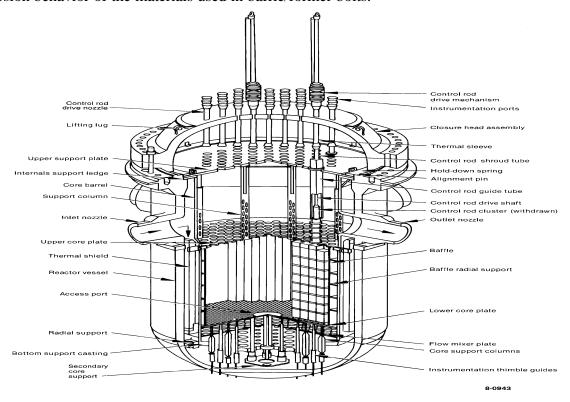


Figure 3-11: PWR core internals, including baffles and formers

The austenitic stainless steels currently used and potential replacement alloys for use as baffle/former bolts or other internals components are being irradiated to doses up to 80 dpa in test reactors. After irradiation, the materials will be subject to tensile tests, fractographic and microscopic examination. Pressurized tube irradiation creep tests are also planned. Additional corrosion tests in PWR environment will be performed on selected materials. The effects of low helium levels (about 30-75 ppm) and high helium levels (about 300-750 ppm) on tensile and corrosion properties will be determined.

The ongoing NEPO projects in this area are:

Project ID: 3-13

Project Title: Mechanical Behavior of Irradiated Structural Stainless Steels

Project Description: Stainless steels experience a wide variety of types of material property degradation caused by exposure to the reactor core operating environment. These deleterious effects are all influenced by the mechanical behavior of the steels and the way it evolves during in-reactor thermal and irradiation exposure. It also affects the behavior during repair processes such as welding. Understanding the mechanical behavior is a critical aspect of understanding the degradation and repair of the reactor internals. Most of the available results were developed in support of fast reactors, which operate at much higher temperatures than LWRs. The work will determine the mechanical behavior of irradiated structural stainless steels under conditions of interest to LWRs and develop models describing that behavior that are needed predict component life, assess the results of NDE examinations and guide the timing of corrective actions.

Project ID: 3-30

Project Title: Irradiation Induced Swelling and Stress Relaxation of PWR Reactor Core Internal Components

Project Description: Utilities that operate Pressurized Water Reactors (PWRs) are concerned with the potential of core shroud components experiencing void induced swelling due to neutron irradiation. The dimensional changes might increase loads at bolted or welded joints and accelerate cracking. The objective of this project is to characterize irradiation induced void swelling and stress relaxation related degradation that could occur in operating reactors, and calibrate and extend models the based on swelling in fast reactors for PWR applications.

3.3.2 R&D Needs for Reactor Internals

Fundamental research on the characterization of changes in material behavior of austenitic stainless steel caused by high levels of irradiation is warranted, to improve the basis for inservice examination, engineering evaluation, mitigation, and repair/replacement of reactor pressure vessel internals. This additional research will involve detailed examination of exposed materials removed from service, along with micro-structural material studies and development of models (e.g., IASCC crack growth models) with features beyond those being considered by the industry in current research programs. Some of this fundamental materials research will also apply to related issues, such as IGSCC and stress relaxation. The broad outline of this additional

research is described in the following paragraphs; more information is provided in the detailed task descriptions in the Appendix.

<u>Fundamental Characterization of IASCC.</u> Through the EPRI BWRVIP program, industry has developed an empirical model that can predict crack growth rates in unirradiated materials as a function of loading and water chemistry conditions. However, irradiation produces significant changes in the microstructure and mechanical behavior of stainless steels and that model is probably only applicable to components such as the core shroud that are exposed to relatively low fluences. Although additional empirical data are needed, testing of irradiated materials is difficult and expensive, and development of an empirically based model for materials subjected to high fluences akin to those developed for unirradiated and low fluence materials is impractical. An improved, physically based understanding of degradation processes and their effects on crack initiation, crack growth rates, stress relaxation, and fracture toughness is necessary.

This understanding is needed for materials exposed to a whole range of neutron fluences ranging from relatively low values (core shrouds in BWRs), to moderately high values (top guides in BWRs), and very high values (baffle bolts in PWRs). There is an ongoing international cooperative effort in this area managed by EPRI and studies proposed here will be coordinated with that work.

<u>Less-Susceptible Materials</u>. Stainless steels show a wide variation in resistance to IASCC. However, it is difficult at present to specify the composition and fabrication conditions that will assure that a material of interest is one of the more resistant materials. EPRI sponsored work has led to the development of a material specification for improved Type 348 stainless steel for thin-section, in-core applications, but at present it is not possible to specify the composition and fabrication conditions that will ensure high resistance for components that are not fabricated from thin sheet materials. Work is needed to identify more resistant materials for such components.

IASCC Weld Repair Methods. When continued structural integrity cannot be ensured, decisions will have to be made about repairing or replacing components. Replacing components is generally an option of last resort due to the high radiation dose rates and costs involved in such an operation. In situ repairs to date have relied on mechanical repairs (clamps, etc.), whose applicability is limited to certain component geometries, because welding of highly irradiated stainless steels is very difficult. There has been substantial work by the industry to develop techniques for underwater welding of stainless steels, leading to the development of industry consensus standards. However, to date such techniques have been successfully applied only for components with relatively low fluence.

During neutron irradiation of stainless steels, helium is formed by neutron reactions with some of the alloy constituents. Helium has a low solubility in metals, and tends to form small clusters or bubbles. At elevated temperatures, these clusters and bubbles can grow and migrate to grain boundaries very rapidly and severely weaken these boundaries. Because conventional welding processes involve high temperatures and result in significant stresses, irradiation-induced helium can severely affect the weldability and the post-weld properties. The DOE fusion reactor

research program has been actively investigating this problem and substantial progress has been made in developing a thorough understanding of the physical processes that cause such cracking. This has led to some promising techniques for solving the problems encountered while welding high fluence materials.

However, in the fusion-related research the focus has been on welding in dry environments and hence the work is not directly applicable to the welding problems of interest for light water reactors. Also, work has been done at Savannah River on this problem. Additional research is needed to develop an underwater welding technique for irradiated stainless steels that is amenable to remote execution, can be used for an actual load-carrying structural repair (rather than simply a superficial covering of a crack), provides adequate inspectability of the repair, and will have ASME code recognition and qualification. Such work will draw on industry experience in remote underwater welding techniques, DOE national laboratory experience in developing welding techniques for irradiated materials in the fusion reactor and defense programs, and DOE national laboratory capabilities to handle highly irradiated materials. Because of the expense and difficulty of working with highly irradiated materials, both experimental studies and analytical studies of the welding of these materials may be pursued.

<u>IASCC/IGSCC Mitigation Methods.</u> High temperature surface annealing of BWR vessel internals surfaces using a plasma heat source can dissolve chromium carbides and eliminate stainless steel sensitization. Associated fast cooling from the annealing temperature will avoid formation of new carbides on the surface. Laboratory results show that such an annealing process can eliminate surface sensitization completely. Further work is needed to develop field portable equipment and techniques.

The high priority projects identified for FY 2001 are continuation of projects 3-13 and 3-30 described in Section 3.3.1. See Volume II for detailed descriptions of these projects.

3.4 Reactor Pressure Vessels

The reactor pressure vessel (RPV) is arguably the most important single safety-related component in a nuclear power plant. Demonstration of RPV integrity (determined principally by its resistance to brittle fracture) is critical to ensure continued, safe plant operation. A typical PWR RPV is shown in Figure 3-12 with the important subcomponents identified. The major issues related to managing plant aging for the RPV are: (1) managing the effects of neutron radiation induced embrittlement for those portions of the vessel (e.g., the beltline region) that are exposed to sufficiently high neutron flux over a sufficiently long time to reduce the fracture toughness of the beltline materials to levels of concern; and (2) managing the effects of primary water stress corrosion cracking (PWSCC) for vessel penetrations, such as those for insertion of instrumentation and control rods. Other plant aging issues associated with the vessel, such as cracking of the vessel interior cladding, fatigue cracking of nozzles, and stress corrosion cracking of closure studs, are of less significance.

The current approach for managing the potential effects of neutron irradiation embrittlement is based on a combination of: (1) periodic in-service inspection (e.g., volumetric examination by

ultrasonic methods), in order to detect and size any significant pre-service or service-induced flaws; (2) beltline material surveillance, using material specimens exposed to an accelerated neutron irradiation environment (i.e., lead specimens), to monitor reductions in fracture toughness; and (3) potential operational controls, as required by regulations. The combination of these three program elements are used to evaluate the fitness of the RPV for continued service.

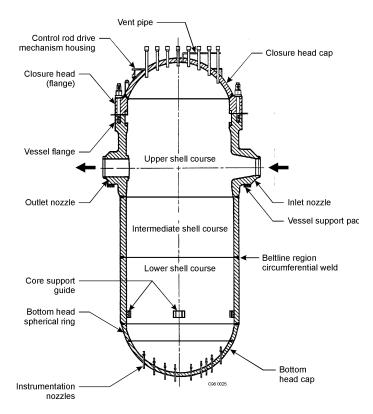


Figure 3-12: Cross section of a typical Westinghouse RPV

The sub-issues of the adequacy of the material surveillance program, the satisfaction of pressurized thermal shock regulations, potential low upper shelf toughness, and potential pressure-temperature operational limits.

The current approach for managing the effects of PWSCC on RPV penetrations primarily involves periodic in-service inspection (e.g., visual examination) to detect leakage from throughwall cracking in Alloy 600 material, plus augmented inspection (e.g., surface or volumetric examination) and other corrective actions to characterize and remediate relevant conditions of concern.

The current approaches to RPV integrity assessment (including NRC regulations and methodologies of standards organizations) were originally implemented incorporating a high degree of conservatism to account for uncertainties. As plants age, operation of the RPV may become unnecessarily restricted due to the conservative nature of existing regulations and assessment procedures. By utilizing advancements in material characterization and integrity assessment technologies, a more realistic determination of RPV integrity can provide the

flexibility required by licensees and still maintain adequate levels of safety against vessel failure. As radiation-induced embrittlement of the RPV increases with plant operation, the continued demonstration of RPV integrity will become a critical factor in the decision whether to pursue additional operation under license renewal provisions. Resolution of RPV integrity issues can result in substantial cost savings through reduced regulatory scrutiny, increased operating flexibility, and extended operation.

3.4.1 Current Reactor Pressure Vessel R&D

Current RPV R&D is directed at reducing the uncertainties in the evaluation of the fitness for continued service. The sound, technical demonstration of these improved evaluation techniques will be a factor in ensuring continued operation.

<u>Reactor Pressure Vessel Embrittlement Management</u>. EPRI developed and maintains databases of information on vessel embrittlement. These serve as a repository for industry data and are an aid to identification of inconsistencies in materials. In addition, EPRI developed guidelines to assist utilities in evaluating uncertainties in material properties.

Industry and the NRC are currently supporting efforts to demonstrate and establish the use of the "Master Curve" approach to the determination of fracture toughness reference temperatures. This testing approach has been standardized by the American Society for Testing Materials. This approach will provide a more realistic assessment of initial RPV fracture toughness and should reduce unnecessary conservatisms associated with RPV integrity assessments.

EPRI has also investigated alternative approaches to the determination of deformation and fracture characteristics of RPV materials. One such technique is the small punch (SP) test. The SP test uses a small "shirt button" disk of material. EPRI has successfully demonstrated its applicability on turbine rotor materials. A joint program is presently underway with CEA (France) to demonstrate the applicability of the SP technique to testing both unirradiated and irradiated RPV materials for application to RPV integrity assessment. At present the SP technique is not universally accepted for use in RPV integrity assessments.

EPRI is currently cooperating with NRC on the NDE of sections of a decommissioned RPV in order to develop more representative generic or postulated characterization of flaw density and distribution. Methodologies to predict flaw distributions based on vessel fabrication techniques and procedures are also being developed. EPRI will pursue a strategy to combine these technologies to develop more realistic estimates of flaw distributions. As part of its revaluation of the pressurized thermal shock rule, the NRC is also undertaking an attempt to define more realistic estimates of flaw distributions.

RPV Pressure and Temperature Limit Optimization. To ensure an adequate margin against fracture, limits are placed on the internal pressure during heating and cooling of the vessel. The allowable pressure-temperature (P-T) limits for reactor heatup and cooldown are defined by Appendix G to Section XI of the ASME Code for the beltline region of the reactor pressure vessel. Recent ASME XI activities based on EPRI and NRC research have focused on developing revisions to Appendix G to allow the use of improved analysis methods with more

explicit definition of the assumptions and the intended margins that are to be included in the analysis.

Irradiation-Induced Changes in RPV steels. In collaboration with CRIEPI (a Japanese nuclear energy consortium), EPRI is supporting fundamental studies of radiation embrittlement of RPV steels using state-of-the-art microstructural techniques to provide insight into the mechanisms that cause radiation embrittlement. NRC is also sponsoring similar fundamental studies at the University of California, Santa Barbara and at the Oak Ridge National Laboratory. An understanding of these mechanisms as reflected through changes in the steel microstructure can be a valuable tool in developing predictive micro-mechanical models. It is expected that this work will lead to improved physically based trend formulas for predicting radiation embrittlement.

Resolution of PTS Issues. An analysis of the Yankee Rowe RPV highlighted the shortcomings of present methodologies for evaluating pressurized thermal shock as outlined in NRC Regulatory Guide (RG) 1.154. The uncertainties associated with the probabilistic fracture mechanics (PFM) and thermal-hydraulics analyses resulted in very conservative estimates of vessel failure probabilities that would have required extensive plant modifications. In response to this issue, EPRI developed an alternative approach to evaluation of PTS, which is intended to more effectively evaluate the impact of corrective measures such as flux reduction, plant modifications, and changes in plant operating procedures on susceptibility to PTS. A plant demonstration of the methodology was completed. In addition, a joint program between EPRI and the NRC is underway to develop a technical justification by December 2001 to revise the PTS screening criteria in 10 CFR 50.61. Advances in probabilistic risk assessment (PRA), probabilistic fracture mechanics (PFM), thermal hydraulics (TH), and overall plant risk assessment are being incorporated into a comprehensive program to establish a technical basis for revising the present screening criteria.

There are no ongoing projects under the NEPO program for this technology area.

3.4.2 Reactor Pressure Vessel R&D Needs

Further research is warranted to reduce the conservatism required in the management of neutron irradiation embrittlement effects and the assessment of the fitness of reactor pressure vessels for continued service. Much of this additional research will rely on fundamental material studies and development of techniques beyond those under current application by the industry. The broad outline of this additional research is described in the following paragraphs. There are several potential approaches to reducing the conservatism in the evaluation of the fitness of the RPV for continued service. One is to reduce the conservatisms in fracture mechanics analyses of vessels during operational transients such as plant heatups and cooldowns or during postulated accidents such as pressurized thermal shock. Another is to better characterize the embrittlement in the material and its effect on the fracture toughness of the RPV. A better characterization can be achieved by an improved understanding of the fundamental mechanisms of embrittlement, by improved techniques to directly measure fracture toughness of the material and by better testing techniques that can make more efficient use of the limited amount of surveillance materials that are available

State-of-the-art microstructural techniques such as atom probe tomography can be used to provide valuable insight into the mechanisms that cause radiation embrittlement. An understanding of these mechanisms as portrayed through changes in the steel microstructure can be a valuable tool in developing predictive micro-mechanical models. A new tool has recently become available, the DOE 7 GeV Advanced Photon Source. This machine provides X ray intensities and energies several orders of magnitude higher than previous sources, which makes it possible, for example, to select out and characterize via X-ray scattering only copper precipitates in material containing a variety of other small-angle scatterers such as carbides and defect clusters.

The fracture toughness of reactor pressure vessels (RPVs) increases with temperature until an upper shelf region is reached. To ensure an adequate margin against fracture, limits are placed on the internal pressure as a function of temperature during heating and cooling of the vessel. The P-T curve, which defines the upper bound of the P-T operating envelope, has been determined using conservative fracture-mechanics-based criteria and analysis methods. Additional research is proposed to develop more realistic P-T curves using more refined analysis techniques and an improved understanding of the behavior of pressure vessel materials under such conditions.

For this technology area, no high priority projects were identified for FY 2001.

3.5 Structures

Maintaining the integrity of reactor containment structures is required for plant and public safety. Schematics of a large dry PWR prestressed concrete containment and a BWR Mark I type metal containment (enclosed in a reactor building) are shown in Figures 3-13 and 3-14, respectively. The in-service performance of reinforced and post-tensioned concrete structures and metal containments in U.S. nuclear power plants has generally been good. However, incidences of age-related degradation have been reported and may increase due to aging of the materials. Comprehensive in-service performance reviews indicate that damage to the concrete structures is primarily in the form of concrete cracking, corrosion of embedded steel reinforcement and liner materials, larger than anticipated loss of pre-stressing forces, and leaching of calcium hydroxide from the concrete. In addition, post-tensioning tendon wires have failed and tendon corrosion inhibitor has appeared on the outside surfaces of several post-tensioned concrete containments. Furthermore, due to lack of direct means of monitoring the integrity of containment basemats, the condition of the concrete in these components cannot be easily established. Some modest corrosion of the metal containments, primarily at the wall-basemat interface and adjoining wall locations, has also been observed.

The need to demonstrate the integrity of aged structures and components in inaccessible areas continues to be a critical issue for license renewal. Although this issue is applicable to all structures/components, for the current operating license period the only regulatory requirements are related to containments. For containments, 10 CFR 50.55a requires an evaluation of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or

result in degradation to such inaccessible areas. For license renewal the NRC has concluded that in some cases the conditions in accessible areas may not always be indicative of the conditions in inaccessible areas.

Effective aging-management programs to demonstrate long-term integrity and reliability of safety-related concrete and metal containment structures requires technologies for detecting and

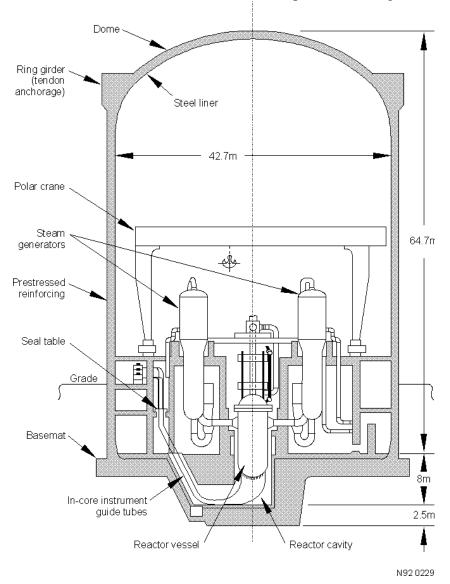


Figure 3-13: PWR prestressed concrete containment

quantifying the extent of the damage and an understanding of the degradation mechanisms. Once a problem is identified, remedial actions can be taken to restore a structure's integrity, enhance its reliability, and extend its service life. For this approach to be effective, developmental efforts need to address the enhancement of concrete technologies in selected areas, as well as improvements in metal corrosion detection capabilities. Computerization of a comprehensive materials property database on long-term concrete and related material

performance will provide a readily accessible knowledge base for use in performing durability assessments and service life estimations. Particular emphasis should be focused on the development of improved and validated service life models and acceptance criteria for service life estimates. A damage model is needed that reflects interactions between degradation mechanisms. Such a model will be useful in performing time-dependent reliability analyses in

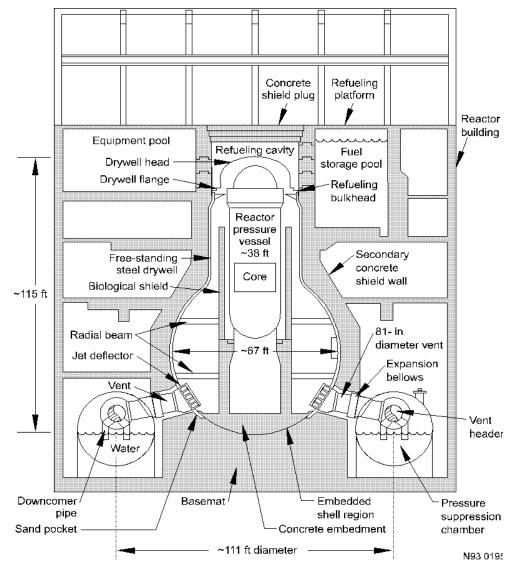


Figure 3-14: BWR Mark I containment (inverted light bulb and torus)

which the failure probability of a degraded component is determined, either at present or at some point in the future. Validation of these analytical tools is considered essential.

User-oriented guidelines need to be developed so that this technology can be reliably used to determine the effect of aging on structural safety margins. Although repair of reinforced concrete structures is performed routinely, supplementary guidance for the repair of nuclear power plant concrete structures should be

developed to address repair strategies and the effectiveness and durability of candidate remedial measures. Currently, there are no NDE techniques capable of providing reliable information about the condition of massive, heavily reinforced concrete structures such as containment basemats. Techniques capable of inspecting inaccessible regions of the containment pressure boundary also need to be developed and demonstrated. Examination of limited in-service inspection data indicates that the use of lift-off loads to estimate pre-stressing forces in post-tensioned concrete containments may overestimate the actual pre-stressing forces. This concern

should be investigated in more detail to determine if this is a significant problem, and if so, to determine the root cause of the problem, its potential impacts on structural margins, and how a more representative estimate of pre-stressing forces can be obtained. The significance of leakage of corrosion inhibitor through cracks in containments should be thoroughly investigated to evaluate its potential effects on the properties of the concrete and overall structural performance.

3.5.1 Current Research and Development

Issues related to the aging of concrete components and its significance relative to useful life of nuclear power plants were initially identified in the 1980s in reports (EPRI NP-2418, EPRI NP-4208, and NUREG/CR-4652) that were prepared for EPRI and NRC. Based on findings and recommendations presented in these reports, a comprehensive review of issues related to the aging of reinforced concrete structures in nuclear power plants was sponsored by NRC. The overall objective of the Structural Aging (SAG) program was to provide NRC with: 1) identification and evaluation of the structural degradation processes; 2) issues to be addressed under nuclear power plant continued service reviews, as well as criteria, and their bases for resolution of these issues; 3) identification and evaluation of relevant in-service inspection and structural assessment programs in use, or needed; and 4) quantitative methodologies for assessing current, and estimating future, structural safety margins. Results of the SAG review were published in a March 1996 report (NUREG/CR-6424) on aging of nuclear power plant concrete structures. Industry reports on Class I structures and PWR containments were also prepared in the early 1990s and published by EPRI. Those reports (EPRI TR-103835 and EPRI TR-103842) addressed license renewal issues and included discussions about effects of concrete aging on long-term performance and safety. EPRI is currently developing an aging management reference manual for condition assessment of concrete structures, and is developing guidelines to assist utilities in satisfying ASME Section XI, IWE/IWL inspection requirements. New projects to address the recommendations provided in the SAG reports for additional R&D in the area of aging management of degraded concrete structures in nuclear power plants have not been initiated.

A companion study to the SAG program is currently being conducted at the Oak Ridge National Laboratory. Under this NRC-sponsored program, issues related to the inspection of aged and degraded metal containments and liners of concrete containments are being identified to 1) understand significant factors relating occurrence of corrosion, efficacy of inspection, and structural capacity reduction of metal containments and liners of concrete containments and make recommendations on use of risk models in regulatory decisions; 2) provide NRC reviewers a means of establishing current structural capacity margins or estimating future residual structural capacity margins for metal containments and liners of concrete containments, as limited by integrity; and 3) provide recommendations, as appropriate, on information to be requested of licensees for guidance that could be used by NRC in assessing the seriousness of reported incidences of containment pressure boundary degradation. A complementary research effort is being conducted at Sandia National Laboratories under NRC sponsorship to investigate and develop analytical methods to determine the effects of corrosion on the ability of metal containments to withstand severe accident conditions. Both of these programs are nearing completion.

The effects of tendon corrosion inhibitor (grease) on the tensile and compressive strengths of concrete used to construct post-tensioned concrete containments have been investigated by NRC. This study focused on the post-tensioned concrete containment at the retired Trojan nuclear power plant. Concrete cores from the containment were removed and tested to characterize and quantify the degradation caused by leakage of corrosion inhibitor through cracks in the concrete. Results of the study at the Trojan nuclear power plant investigating tendon sheathing filler migration have been presented in an NRC contractor report (NUREG/CR-6598)

Technical information, degradation mechanisms, and field experience related to the management of aging damage to light water reactor metal containments were presented in an NRC contractor report (NUREG/CR-5314, vol. 5) issued in 1994.

Brookhaven National Laboratory is conducting an NRC research program on age-related degradation of structures and passive components at nuclear power plants. Instances of aging degradation have been collected and evaluated. The evaluations, trending analyses, available technical information, and risk significance of aging effects were utilized in a scoping study to identify those structures and passive components that warrant further detailed evaluation. The scoping study concluded that masonry walls, flat bottom tanks, anchorages, concrete structures and buried piping warrant further detailed evaluation. Future work will seek to develop methodologies that could be used to quantify the impact of age-related degradation of structures and passive components on plant risk.

There are no ongoing projects under the NEPO program for this technology area.

3.5.2 Research and Development Needs

Research and development required to understand the effects of aging on the future performance of concrete and metal safety-related structures in nuclear power plants needs to focus on key areas covering material properties and performance modeling; structural component inspection, assessment, and remediation; and time-dependent reliability service life estimations. These efforts should be integrated into a comprehensive research program that is divided into tasks that address the following topics: 1) development of improved and validated service life models and acceptance criteria for service life estimates; 2) preparation of supplementary guidance for the inspection, assessment, and repair of nuclear power plant concrete structures; 3) development and demonstration of NDE techniques for inspecting concrete and metal components and inaccessible regions of the containment pressure boundary; and 4) development and evaluation of time-dependent reliability service life estimates.

Selected research and development activities should be initiated to establish the models, methodologies, and tools necessary for effective aging management of concrete structures in nuclear power plants. Development of methodologies and models that assess the current condition of degraded concrete components, estimate the extent of damage caused by corrosion, freeze-thaw exposure, and alkali-silica reactions, and predict remaining life is important. For aging-management programs to be effective in demonstrating long-term integrity and reliability of safety-related concrete structures, technologies for detecting and quantifying the extent of damage that has occurred, and for understanding the effects of degradation mechanisms on

structural response, should also be developed. Candidate NDE techniques that could be evaluated for use in establishing the condition of degraded concrete components and inaccessible portions of containment pressure boundary components include microwave sensors, ultrasonic techniques, guided plate waves, magnetostrictive sensors, aperture imaging, and electromagnetic acoustic transducers. In order to optimize in-service inspection and maintenance strategies based on risk parameters, a time-dependent reliability methodology should be developed. Once developed and validated, such a methodology would be useful in providing reliability-based future condition assessments.

For this technology area, no high priority projects were identified for FY 2001.

3.6 Piping Systems

This technology area focuses on methods for managing service-related degradation in nuclear plant piping systems. In general, operating experience with nuclear power plant piping has been very good over the last 40 years of commercial nuclear power plant operation. The primary degradation mechanisms have been flow-accelerated corrosion and fatigue. As a result of the Surry pipe break in 1985, significant industry efforts were undertaken to understand the mechanism of flow accelerated corrosion and to develop and implement effective tools for managing this degradation mechanism.

The tasks in this area deal primarily with the development of methods and tools to effectively manage fatigue. The primary fatigue issues are the effects of non-design basis transients and the effects of reactor water coolant environment on the fatigue life of piping components. A schematic of a Westinghouse PWR reactor coolant system, with locations of relatively high fatigue usage identified, is shown in Figure 3-15.

3.6.1 Current R&D on Piping Systems

Thermal Fatigue

The EPRI Materials Reliability Program has initiated a Thermal Fatigue Issue Task Group (TF ITG) to provide utilities with methods and tools to manage thermal fatigue which may result from cyclic thermal stratification in Class 1 piping systems attached to and unisolable from the reactor coolant system pressure boundary. Several TF ITG products are currently nearing completion. An *Interim Thermal Fatigue Management Guidelines* document has been prepared and is undergoing final review by the ITG. The *Interim Guidelines* provide assessment and inspection recommendations for detecting potential fatigue damage that may be occurring in normally stagnant lines attached to the reactor coolant system. The *Guidelines* focus on those locations where multiple cases of cracking have occurred in domestic and foreign PWRs and which may not currently be part of an augmented inspection or other fatigue management program. Upon approval by the ITG, the *Interim Guidelines* will be submitted to the MRP Issues Integration Group for review and comment by all MRP utilities. It is intended that the *Interim Guidelines* will be used by PWR utilities on a voluntary basis.

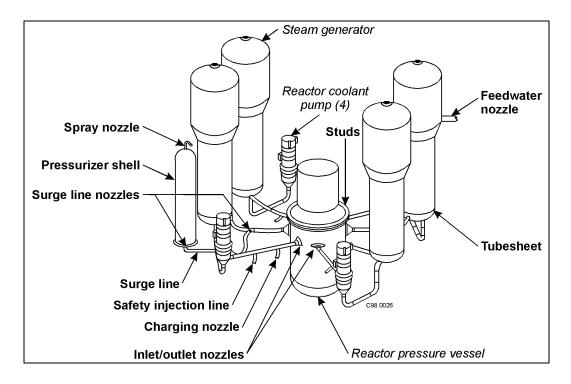


Figure 3-15: Schematic of Westinghouse PWR reactor coolant system with locations of relatively high fatigue identified in bold print

A study of various technologies to detect thermal fatigue cracks in small diameter piping (less than 4 inches in diameter) has been completed and the final report, *NDE Technology for Detection of Thermal Fatigue Damage in Piping* has been approved by the ITG and is currently being published by EPRI. The report describes the evaluation results of candidate NDE technologies for thermal fatigue crack detection and recommends guidance for NDE examiner qualification. Both manual pulse-echo ultasonics and conventional radiography are capable of detecting thermal fatigue cracks. A *Computer Based Training Course for Ultrasonic Examination of Thermal Fatigue Cracking* is currently being developed to train experienced UT examiners to detect and characterize thermal fatigue cracks. After Beta testing by several TF ITG utilities, the CBT Courseware should be available to MRP utilities in December, 2000.

The TF-ITG has also completed a project to document industry operating experience for thermal fatigue in RCS attached piping. Operating experience data of failures due to thermal stratification and cycling that have occurred worldwide has been collected. In addition, the results of instrumenting susceptible lines to measure thermal stratification were collected from domestic PWR plants. Information on examination results, valve leakage, unusual events, and modifications to mitigate thermal fatigue were also collected. The purpose of this operating experience review is to gain an understanding of the quantity and severity of thermal fatigue damage that has actually occurred. By reviewing the failure locations, mechanisms, monitoring results, and corrective actions, more effective tools that address the root causes of actual failures can be developed. The operating experience data collected has been assembled into two databases: major leak events in domestic and foreign plants, and precursors to leakage, such as thermal monitoring results and observed anomalies in domestic plants. The final report entitled

Operating Experience Regarding Thermal Fatigue of Unisolable Piping Connected to PWR Reactor Coolant Systems is currently being reviewed by the TF ITG.

Finally, the Thermal Fatigue Issue Task Group in cooperation with the EPRI Fatigue Pro Users Group conducted an *International Conference on Fatigue of Reactor Components* in Napa, California on July 31-August 2, 2000. The Conference, co-sponsored by the OECD Nuclear Energy Agency and the USNRC was intended to provide a forum for the technical discussion of various fatigue issues that affect the integrity and operation of light water reactor components. The conference was well attended with approximately 90 participents from 12 countries represented. The primary focus of the conference were the issues of thermal fatigue and fatigue reactor coolant environmental effects. The proceedings of the conference will be published as an EPRI MRP report later this year.

Reactor Water Environmental Effects

Laboratory fatigue tests in a reactor water environment indicate that the fatigue life in water is significantly shorter than in an air environment. There is a concern that the ASME fatigue design curves based on laboratory tests in an air environment, with appropriate safety margins, may not have the conservatism originally intended. NRC has concerns that fatigue failures will increase during the license renewal period (after 40 years) as a result of reactor water environmental effects.

License renewal applicants have to consider reactor water environmental effects at a sample of Class 1 locations for the license renewal term in order to receive a renewed license near-term. These sample locations are the locations defined in NUREG/CR-6260 for a specific NSSS design. The influence of reactor water environment on these sample locations will be accounted for in the process licensees decide to take so that potential fatigue damage does not jeopardize the ability of these plant component locations to perform their intended functions.

Guidelines for addressing fatigue reactor water environmental effects in a license renewal application are being prepared to assist utilities in meeting this license renewal requirement and minimizing the amount of plant specific work required.

The ongoing NEPO project in this area is:

Project ID: 3-24
Project Title: Fatigue

Project Description: In the initial licensing period, very conservative assumptions on stresses, fatigue strength reduction factors, and cyclic histories could be made and still demonstrate that fatigue damage was within acceptable limits. In the license renewal period it is increasingly difficult to take this approach. On the other hand, there are also instances where the current fatigue design approach may be nonconservative. Laboratory data demonstrate that, in some cases, LWR coolant environments can have significant detrimental effects on fatigue life. Operating experience has shown that there are thermal fatigue transients that were not properly captured in the original fatigue design basis. This work will help provide cost effective methods of evaluating the cyclic life of nuclear components, including the effects of reactor coolant

environment, based on the safety margins of the ASME Code and provide utilities with appropriate "tools" to manage fatigue effects.

3.6.2 R&D Needed on Piping Systems

Thermal Fatigue

Methodologies are needed to predict where and under what conditions thermal cycling may occur in PWR piping system components. Thermal cycling results from cyclic thermal stratification or cyclic mixing of hot and cold fluids. These methods will allow utilities to minimize the potential for pipe cracking due to thermal fatigue.

Reactor Water Environmental Effects

Additional test data and analyses are needed to determine the timing and real significance of fatigue reactor water environmental effects, and to develop the appropriate aging management programs for long term plant operation.

The high priority project identified for FY 2001 is a continuation of project 3-24 described in Section 3.6.1. See Volume II for a detailed description of this project.

3.7 Generic R&D

Much of the current research and development on aging management is component specific – such as the management of irradiation embrittlement effects on low-alloy ferritic steel reactor pressure vessel materials and the management of irradiation-assisted stress corrosion cracking of austenitic stainless steel reactor vessel internals. In some cases, however, R&D is more generic in nature, extending to vessels, tanks, piping, pumps, valves, internals, and balance-of-plant components. The crosscutting tasks described here are more generic in nature and affect all types of components. For example, some of the current R&D is focused on materials information and analytical assessment tools that are applicable to a variety of components. Other current generic R&D relates to work force quality and the application of modern risk assessment tools and technologies to capture the risk impact of aging. All of these topics have potential impact on the license renewal process.

In addition, with the availability of components and structural materials from plants that have been retired from service, generic data on the effects of aging through examination of components and structures can be obtained. This information can provide a useful, realistic assessment of aging effects on important plant components and will "benchmark" data on the effects of aging obtained from accelerated laboratory or test reactor studies.

Another example of a crosscutting task involves long-range research needed to develop and demonstrate methods for direct measurement and assessment of the early precursors of material

degradation, such as in-grain or grain boundary precipitates and severe dislocation patterns in metals.

3.7.1 Current Generic Research

Reduce Plant Inspection Costs. The objective of a current EPRI project is to reduce plant inspection costs through the establishment of realistic NDE requirements and the introduction of advanced technology. This project supports integration of relevant EPRI research into the activities of industry groups such as the ASME Boiler and Pressure Vessel Code Group, ASME Pressure Vessel Research Council and NEI. Using EPRI results, ASME code rules and acceptance criteria will be developed and implemented for reduced RPV reference flaw size assumptions, risk-based inspection, environmental fatigue effects, optimized stress intensification factors, and flaw tolerance evaluations.

Risk-informed approaches to NDE are being developed for inspection applications, to optimize them and make them more cost-effective. New rules for selecting in-service inspection sites are being developed by ASME, the utility industry, NRC Office of Research, and EPRI. The sites selected under the new rules may not have adequate nondestructive inspection technology available to properly implement risk-informed inspections. This project provides technical input during development and implementation of risk-based inspection criteria. The 1998 focus is on supporting pilot plants and developing an NDE reliability model.

Another area of improvement under consideration is flow-assisted corrosion (FAC) inspection techniques, which provide a measurement of remaining pipe wall thickness. Eighty percent of current FAC inspection costs are associated with removal and reinstallation of piping insulation. An improved technology involving low-strength radiography for measuring wall thickness through insulation is being pursued by EPRI.

Another issue being addressed in this project is the examination of dissimilar metal welds (e.g., Alloy 182 safe-end welds). Some of those welds have such complex configurations that accurate interpretation of the inspection signals and photographs has been extremely difficult.

Advanced NDE Technology. Utilities need cost-effective and reliable inspection methods and analysis programs that can be integrated easily with structural and lifetime evaluations of nuclear power plant components and systems. This EPRI project focuses on the development of improved NDE hardware, software, databases, and methods. The deliverables address development of NDE technology to solve existing and projected integrity management issues while maintaining an element of innovative research to solve future problems. Nondestructive methods for addressing material properties and damage prior to cracking are included.

<u>Materials Handbook</u>. This research and development activity at EPRI is aimed at a materials handbook for nuclear plant pressure boundary applications that can be used as the primary source document by utility materials engineers facing questions regarding repairs, replacement materials, trouble-shooting failures, failure analysis, and in-service inspections. While much of the desired information is available in other publications, it has not been compiled in one place with emphasis on factors important to nuclear power plant pressure boundary applications. A

related current research activity at EPRI is aimed at providing guidance on the useful service life of valve stem and bolting alloys that are susceptible to thermal embrittlement. Service failures have resulted from embrittlement of components in PWRs, possibly aided by stress corrosion cracking.

Equipment Assessment & Maintenance Technologies. The cost of operating and maintaining nuclear plants has created a serious challenge to the competitiveness of current and future plants. Cost-effective operations and maintenance (O&M) programs are essential to ensure the continued use of nuclear energy. The success of O&M programs depends on technologies that limit O&M activities, lead to less costly approaches to performing required work, deal with the problems of aging plants, make better use of human resources, provide more cost-effective instrumentation and controls, and support decisions to continue plant operation.

Utilities have reduced O&M costs by controlling staff size, improving plant performance, eliminating unnecessary activities, and applying least-cost options for required actions such as maintenance, inspection, surveillance and testing. Utilities will need to rely on new tools and continuous improvements in equipment reliability, assessment methods, and repair techniques to sustain the downward trend in O&M costs.

A current EPRI project addresses a broad set of plant and equipment reliability, assessment, and repair issues. It is producing specific solutions to these significant O&M cost drivers and is exploiting opportunities with cost-reduction potential. The deliverables will generally consist of 1) incremental design improvements to high-cost impact equipment which will significantly increase reliability, maintainability, performance and service life; 2) innovative lower cost alternatives for required maintenance, condition assessment, surveillance and testing; 3) defensible technical bases for justifying equipment operability and optimizing maintenance frequencies; and 4) technologies that will improve station output. This project includes major piping, valves, rotating equipment, stationary batteries, heat exchangers, and concrete structures, addressing many current and emerging industry issues. Strong coordination and resource leveraging with other organizations, both within and outside EPRI, is being maintained on this project.

Work Force Quality. The supply of qualified inspectors and their level of training and qualification is a concern being addressed by a current EPRI activity. Approximately 30 percent of the EPRI NDE program is directed to training of utility NDE inspectors. In spite of recent advances, training and qualification of NDE personnel continue to be characterized by high costs and low effectiveness. High costs are associated with the labor required to develop and implement training and qualification materials and for operators to reach acceptable proficiency levels. To meet these challenges, EPRI is developing improved procedures to increase reliability, validity, and effectiveness while also reducing costs. NDE training and qualification requirements that can benefit most from improvement will be identified. Interventions based on technologies most applicable to improving performance will be developed and tested. The methods and interventions being developed will support training and qualification programs, such as the Performance Demonstration Initiative.

The ongoing NEPO projects in this area are:

Project ID: 3-27

Project Title: Assessment of Aging Effects on Components and Structures from Nuclear Power

Plants

Project Description: Materials and components that have been in service in operating reactors will be acquired to be used for comparison with laboratory aged materials, to validate models for aging effects, and to validate nondestructive examination methods. Obtaining 40-year equivalent aging data to support plant licensing in laboratory testing requires accelerating the aging process, e.g., using artificially high temperatures or radiation dose rates and extrapolating short duration test data. These results need to be "benchmarked" against components and materials that have aged under prototypical operating conditions. In some cases, e.g., steam generators or electrical cables, the components of interest may become available because they have been replaced. In others they can be obtained from plants which have been retired from service. Opportunities to obtain this information in the past have been lost due to 1) the unavailability of resources, 2) the lack of a systematic evaluation of the needs for component retrieval and assessment, and 3) the relatively short "window of opportunity" to obtain such components before they are sent for disposal at a waste storage site or otherwise rendered inaccessible.

Project ID: 3-3

Project Title: Overcoming Solubility Limitations to Zinc Additions in Pressurized Water Reactors (PWRs)

Project Description: Primary Water Stress Corrosion Cracking (PWSCC) is an increasingly problem in steam generators. Zinc additions to the primary system are expected to reduce the frequency of such cracking. EPRI and Westinghouse jointly conducted trials of zinc addition at two PWRs: Farley-2 and DCPP-1. These trials indicate, however, that there is a potential for zinc to precipitate out on the fuel. The extent of the precipitation depends on the degree of solubility of ZnO, zinc oxide. Potential fuel integrity implications of this deposition have driven these plants to reduce zinc levels. At the lower levels, the PWSCC benefit is significantly reduced. This work will determine the solution thermodynamics of the zinc/zinc-oxide (Zn/ZnO) system at high temperature in order to permit maximum zinc additions without ZnO precipitation.

3.7.2 Generic Research Needs

Section 3.7.1 identifies a number of issues and related work being pursued by EPRI. The scope of the currently planned work is limited and there is a need for expanding the scope and performing R&D to address these issues effectively and completely. For example, on the issue of work force quality, technology could be devised that creates a virtual environment, permitting an ultrasonic testing operator to conduct practice examinations while directly observing the entire process – effects of transducer manipulation, sound paths in the material, effects of geometry, displayed signals, and interactions. The immediate and dramatic feedback of the

results of examination strategies, procedures and actions might significantly reduce the time required for operators to become proficient, and would provide a better foundation for skill retention.

There is a need to obtain data on the effects of aging through examination of components and structures that have aged under service conditions. Components removed from nuclear power plants are needed to provide a useful, realistic assessment of aging effects on important plant components and to "benchmark" data on the effects of aging obtained from accelerated laboratory or test reactor studies. In addition to providing information on the effects of aging. these components would also be useful in providing information on promising new techniques and devices for nondestructive material damage assessment (e.g., RPV "embrittlement meter" and direct measurements of fatigue damage). These components can be obtained from a number of sources, such as power plants that have been retired from service or components that have been replaced (e.g., steam generators). Opportunities to acquire such components for analysis in the recent past have been lost (e.g., Shippingport, Yankee Rowe, numerous steam generator replacements) due to 1) the unavailability of resources, 2) the lack of a systematic evaluation of the needs for component retrieval and assessment, and 3) the relatively short "window of opportunity" to obtain such components from the time a plant is shutdown or a component is replaced to the time the components are sent for disposal at a waste storage site or otherwise rendered inaccessible.

Long-term research is needed to develop and demonstrate additional methods of detection of material degradation, such as in-grain or grain boundary precipitates and severe dislocation patterns. The development of methods for direct measurement and assessment of early precursors requires the continued development of physically-based material models so that the characteristics of susceptible materials can be identified and quantitative predictions of damage or condition assessments as a function of service exposure can be made. Development of in situ examination techniques that can detect the physical manifestations of material degradation, correlating with the physically-based material models will also be needed. In-situ techniques for examining material composition, chemistry, and microstructure, leading to the identification of early indicators of material degradation, will first be demonstrated in the laboratory (or hot cell) and then validated through destructive examination of components removed from service.

The determination and quantification of nuclear plant risks from aging concerns will need to be addressed as plant owners prepare for license renewal. A comprehensive probabilistic safety assessment (PSA) package will allow plant owners and operators to prioritize maintenance, inspection, and replacement activities taking aging effects into account, and to address regulatory concerns related to public risk. Development of such a PSA package is another generic R&D need related to power plant aging.

There are two high priority projects identified for FY 2001 in this area. The first is a continuation of project 3-27 described in Section 3.7.1. The second project is identified below. See Volume II for detailed descriptions of these projects.

Project ID: 3-29

Project Title: Motor Rewind Insulation System Development and Qualification for Harsh

Environment

Principal Objectives: To design and qualify motor insulation systems for use in harsh

environment applications